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Notch Ductility and Tensile Property Evaluation of the PM-2A Reactor Pressure Vessel

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ABSTRACT

Following the pressurization-to-failure testing of the PM-2A reactor pressure vessel, several sections of steel were removed from the vessel wall in a region adjacent to the artificial defect. Charpy V-notch and tension test specimens machined from one of these sections have been evaluated. The irradiated-condition 30 ft-lb transition temperatures for the 1/4-thickness (nearest to the core) and 3/4-thickness locations in the vessel wall were +115°F and +55°F, respectively, for measured fission-spectrum fluences of 7.3 and 4.0×10^{18} n/cm^2 (> 1 Mev). The 1/4-thickness properties and fluence most nearly represented those at the tip of the artificial defect. The 0.2% yield strength for the 1/4-thickness location was 97,620 psi at -20°F (failure temperature) and 92,200 psi at +72°F (temperature at time of acid-sharpening treatment of artificial defect). Significant uniform elongation, reduction of area, and elongation per 1 in. were retained by the steel. An assessment of the stress, temperature, and flaw-size conditions for the PM-2A failure, as indexed by the irradiated-condition mechanical properties, indicates that the failure is in agreement with the generalized fracture analysis diagram.

PROBLEM STATUS

This is an interim report on one phase of the problem; work is continuing.

AUTHORIZATION

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NOTCH DUCTILITY AND TENSILE PROPERTY EVALUATION OF THE PM-2A REACTOR PRESSURE VESSEL

INTRODUCTION

The PM-2A reactor was built by Alco Products, Inc., for use at the U.S. Army Polar Research Development Center, located beneath the ice cap at Camp Century, Greenland. The portable, medium-power (10 Mwt), pressurized light-water reactor was compact and therefore relatively simple to transport and erect. One penalty of such compactness, however, was the relatively high neutron flux (about 1.9×10^{11} neutrons per square centimeter per second (n/cm^2-sec) of energies greater than 1 Mev) at the inner edge of the carbon-steel vessel wall. This relatively high flux impinging on a pressure vessel of steel sensitive to radiation-induced embrittlement had caused a significant elevation of the vessel brittle-ductile transition temperature at the time of terminal shutdown of the plant, July 9, 1963.

Because it was felt that much could be learned from the PM-2A vessel, it was shipped to the National Reactor Testing Station in Idaho for nondestructive examinations and a series of pressurization tests under controlled conditions to effect a brittle fracture of the pressure vessel. Under the direction of the Bettis Atomic Power Laboratory (BAPL), the vessel was examined nondestructively for dimensions, cracks or defects, hardness, and neutron fluence. Results of this Phase I program have been reported by Monahan and Halpine (1). Detailed calculations and nondestructive measurements of the neutron fluence of the PM-2A vessel have been described by Shure and Oberg (2). The Phase II program, under the direction of BAPL, was concerned with the actual pressurization-to-failure of the vessel and has been reported in detail by Monahan and Walker (3).

Following the Phase II pressurization tests actually carried out by the Idaho Nuclear Corporation (INC) (4), the U.S. Atomic Energy Commission (AEC) asked NRL to direct the effort for laboratory evaluations of the mechanical properties of the vessel steel in the region of failure, since NRL had performed extensive studies of the PM-2A-type material. Specimen requirements for these evaluations were met by the removal of steel by INC from the PM-2A vessel in a horizontal region adjacent to and vertically overlapping the artificial defect (Fig. 1). Wedge Opening Load (WOL) fracture-mechanics specimens (to be evaluated by Battelle Memorial Institute, Columbus, Ohio) were machined by INC from the first slab which was directly adjacent to the defect; drop-weight, Charpy-V, and tensile specimens were machined from the next two sections. Phillips Petroleum Company (PPC) obtained the final section of steel for ultrasonic examinations. This report presents the results of evaluating Charpy V-notch and tensile specimens and the associated neutron dosimetry data obtained from the PM-2A vessel, along with the results of a calculated neutron spectrum analysis through the vessel wall thickness.

MECHANICAL PROPERTIES

The pressure vessel of the PM-2A reactor was fabricated from two 2.4-in.-thick ring forgings of A350-LF3 steel. The chemical composition of the upper ring forging (core region), determined by the reactor builder, Alco Products, Inc., and by INC after irradiation, are given in Table 1, along with the heat treatment of the forgings. Similar data for a sister forging produced by Alco from the same heat as the vessel forgings are

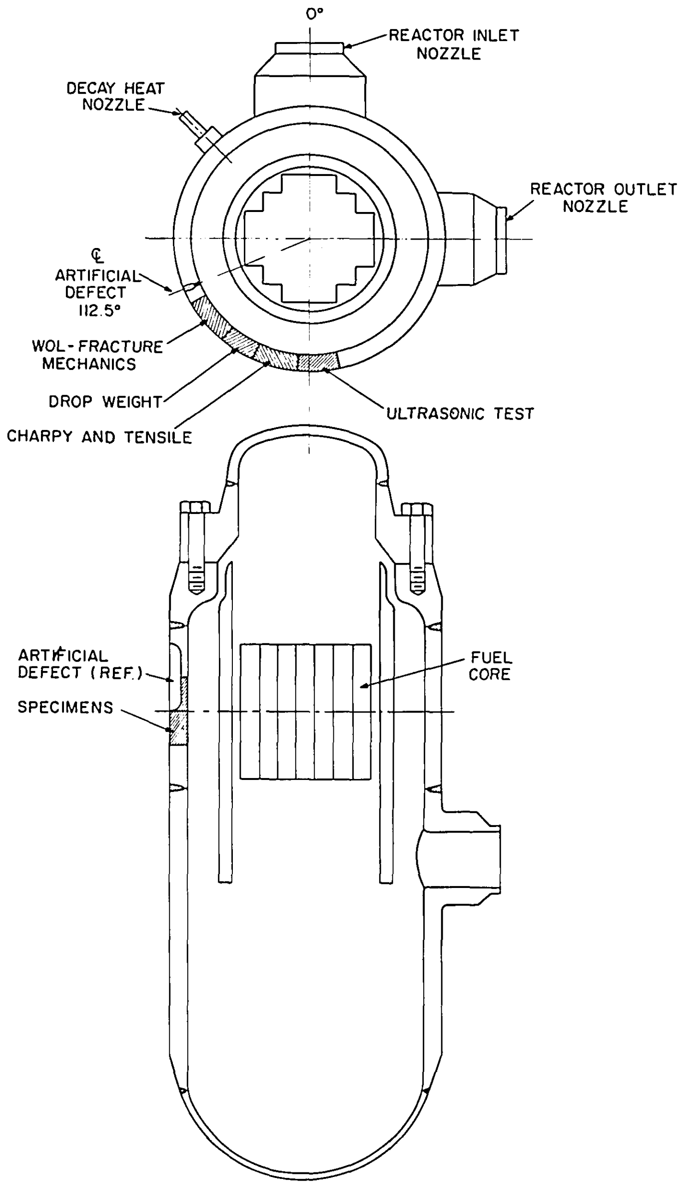

Fig. 1 - PM-2A reactor vessel showing the relative locations of the artificial defect, fuel core, and sections of the vessel wall subsequently removed for metallurgical specimen preparation

Table 1
Chemical Composition and Heat Treatment of 2.4-in. A350-LF3 Ring Forgings

Evaluating Laboratory	Composition (wt-%)										
	C	Mn	Si	P	S	Ni	Cr	Mo	V	Cu	Fe
	PM-2A Forging*										
Alco	0.14	0.52	0.25	0.031	0.032	3.28	0.04	0.05	0.04	—	—
INC†	0.17	0.53	0.24	0.032‡	0.03	3.36§	—	—	0.035	0.14	95.58#
	Sister Forging**										
NRL	0.14	0.52	0.25	0.031	0.032	3.28	0.04	0.05	0.04	—	—

*Heat treatment for PM-2A forging was as follows: forged in the temperature range of 1700°F min to 2250°F max. Normalized at 1550°F; water quenched; tempered at 1200°F; stress relieved at 1150°F.
†Radiochemical separation after irradiation of vessel.
‡±0.002%.
§±0.15%.
#±1.05%.
**Heat treatment for sister forging was as follows: normalized at 1600°F; water quenched; tempered at 1250°F; stress relieved at 1150°F.

also shown (Table 1) for comparison purposes. The sister forging was used for the experimental irradiation program on the PM-2A steel.

Notch Ductility Data

The nil-ductility transition (NDT) temperature* of the upper ring forging of the PM-2A-reactor pressure vessel and the preservice Charpy-V-notch ductility characteristics were not determined at the time of manufacture. As a first approximation of these values for the unirradiated vessel condition, the NDT and Charpy-V values determined for the sister forging were used. Drop-weight specimens which are usually employed for NDT determinations have been machined from the PM-2A vessel (Fig. 1) but have not been tested pending successful development of a pressed-notch crack starter in lieu of a brittle-weld crack starter. It is felt that welding will anneal some of the damage, thus yielding a false, tougher steel condition.

The reactor vessel Charpy-V specimens were evaluated in two groups. Those of the inner radius (nearest the core) were taken at the 1/4-thickness (t) location of the vessel

*A differentiation must be made between a nil-ductility transition (NDT) temperature increase and a transition temperature increase. An NDT temperature increase implies that drop-weight specimens of a material have been tested to establish the NDT temperature directly. Charpy V-notch results for the same materials may then be indexed to this temperature by selecting the corresponding energy value ("fix point" of the Charpy curve) which best represents the NDT temperature. This level can then be used for the determination of the neutron-induced NDT increase. A transition-temperature increase refers to the method of simply selecting an arbitrary energy level (preferably based on an average correlation point representative of the NDT for that class of steels). Then, the increase in temperature of the reference value is determined by translating along that energy level to the curve for the irradiated condition.

wall (Fig. 2), and the outer-radius specimens were taken from the 3/4-thickness (t) location. The inner-radius (1/4 t) specimens of both the longitudinal (notch perpendicular to the wall thickness) and transverse orientation (notch parallel to the wall thickness) showed the same response (Fig. 3). Similarly, the outer-radius (3/4 t) specimens of both the longitudinal and transverse orientations also showed no discernible difference between the two test orientations (Fig. 3). The transition temperature increase, referenced to the initial properties of the sister forging and indexed to the postirradiation Charpy-V 30-ft-lb energy level, was 195°F for the inner-radius specimens, placing the irradiated-condition transition temperature at +115°F. An increase of 135°F was described by the outer-radius specimens, corresponding to a postirradiation Charpy-V 30 ft-lb transition temperature of +55°F. These data are presented schematically in Fig. 4.

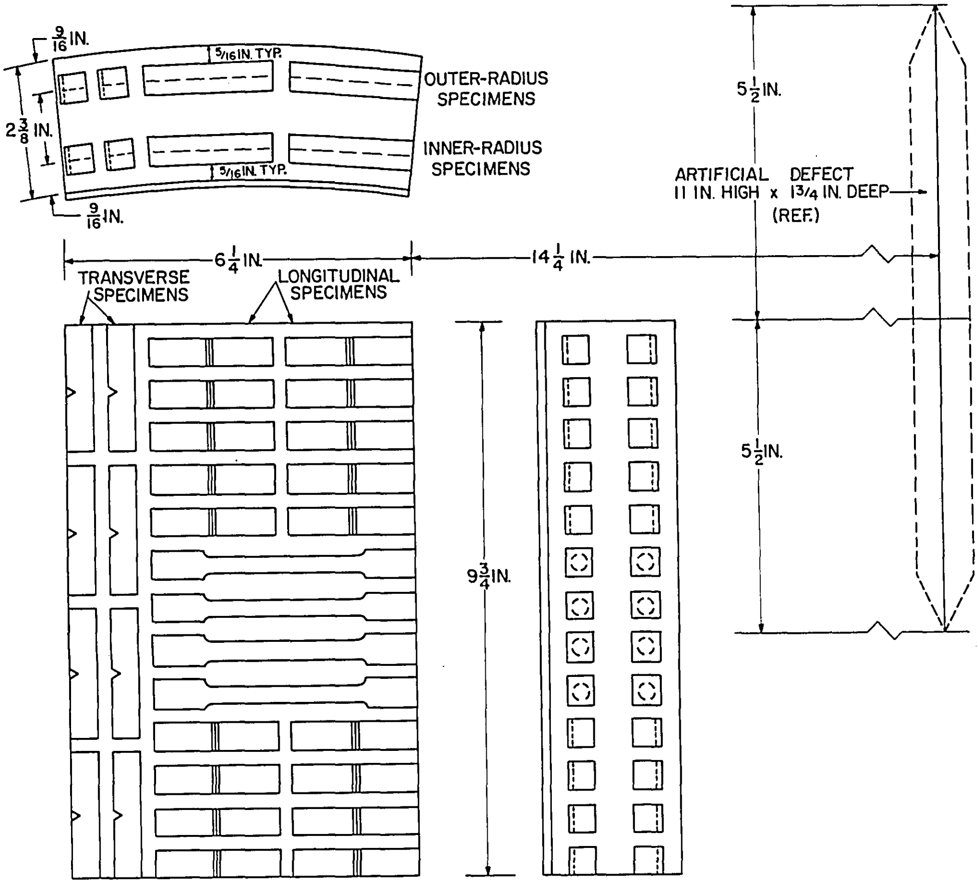

Fig. 2 - Location and orientation of Charpy V-notch and tension test specimens within the parent section of the pressure-vessel steel of the PM-2A ring forging

Based on these data, a Charpy-V 30 ft-lb transition temperature of +160°F would be projected for the inner edge of the A350-LF3 steel vessel wall at a neutron fluence of 1.0×10^{19} n/cm^2 > 1 Mev (indicating an instantaneous flux of 1.94×10^{11} n/cm^2-sec > 1 Mev), assuming a fission-spectrum neutron distribution. The Charpy-V data of the as-irradiated condition are summarized in Table 2.

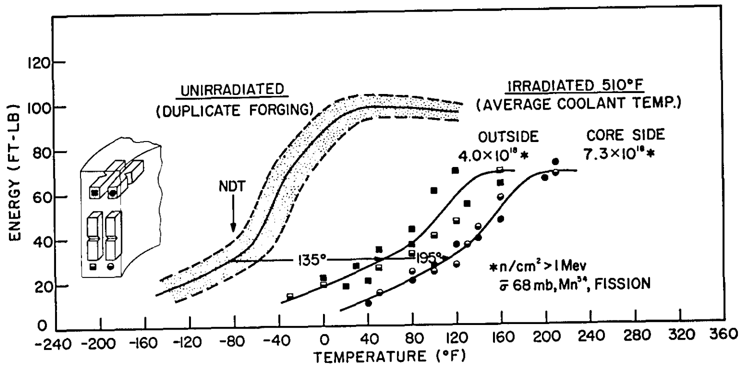

Fig. 3 - Notch ductility characteristics of PM-2A ring-forging steel (A350-LF3). Irradiated specimens were taken from two thickness locations in the ring forging and from both longitudinal and transverse orientations. The unirradiated-condition Charpy-V characteristics are those of a sister forging made from the same vessel heat.

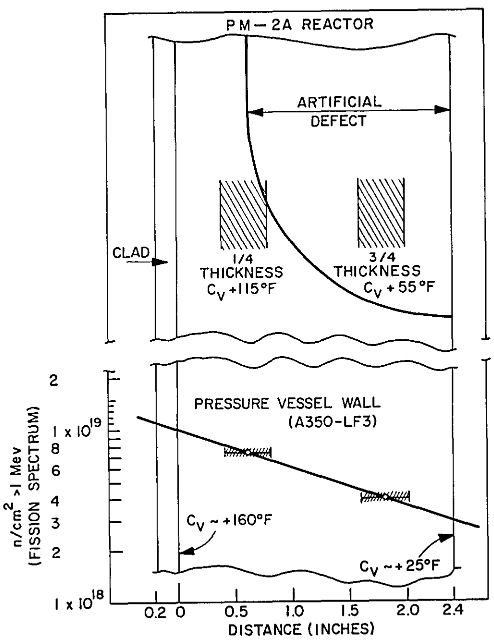

Fig. 4 - Neutron fluence gradient and irradiated-condition Charpy-V 30 ft-lb transition temperatures for locations across the thickness of the PM-2A-reactor vessel wall. The volume of material occupied by the Charpy-V specimens is shown in relation to the artificial defect. The bands on the neutron fluence points indicate the range of material averaged by the measurement.

Table 2
Irradiated-Condition Notch Ductility of the Pressure-Vessel Wall of the PM-2A Reactor (2.4-in. A350-LF3 Ring Forging)

Orientation	Test Temp. (°F)	Energy (ft-lb)	Shear (%)	Rockwell Hardness
1/4 t location; neutron fluence of 7.3×10^{18} n/cm^2 > 1 Mev (fission)				
Longitudinal	80	36	50	—
	0	22	20	—
	50	34	40	92.5
	130	54	100	94.5
	20	18	25	93.5
	40	20	25	—
	160	63	100	—
	120	69	85	94
	30	27	35	—
	80	43	50	93.5
	-30	30	15	93.5
	100	60	65	92
Transverse	80	32	45	—
	-30	15	10	94
	50	26	25	—
	0	19	15	—
	100	40	70	94
	120	46	60	93
	40	39	35	—
	160	69	100	—
3/4 t location; neutron fluence of 4.0×10^{18} n/cm^2 > 1 Mev (fission)				
Longitudinal	200	65	95	95.5
	80	20	30	—
	100	27	35	95
	160	47	55	—
	20	36	40	—
	50	27	30	94
	40	10	20	93
	140	39	50	95
	210	72	100	—
Transverse	130	36	55	95
	120	27	40	—
	50	15	15	94.5
	80	24	30	—
	140	44	60	95.5
	100	24	35	96.5
	160	57	80	—
	210	68	100	—

The Charpy-V specimens not broken for development of the as-irradiated condition of the PM-2A vessel steel were postirradiation heat-treated at 650°F for 168 hr (Table 3). The data points are shown in Fig. 5 as referenced to the irradiated-condition curves of Fig. 3. Since the recovery of preirradiation properties was small for this 650°F heat-treatment condition, the remaining specimens were subjected to an additional 168 hr at 750°F. As can be seen, this combination of heat treatments resulted in very close correspondence with the unirradiated-condition band for the Charpy-V data of the sister-forging, suggesting essentially complete recovery. For comparison purposes, a test reactor irradiation of sister-forging specimens was conducted at 510°F, the results of which showed similar recoveries of about 25 to 30% of the unirradiated-condition properties for a 168-hr, 650°F heat treatment and essentially 100% for the additional 168-hr, 750°F anneal. These recovery data also compare well with other postirradiation heat-treatment data developed from evaluations of the same sister forging (5). Thus, in the absence of preirradiation Charpy-V values for the PM-2A vessel steel, these results give confidence to the use of -80°F as the initial 30-ft-lb value.

Table 3
Postirradiation-Annealed Condition of the Pressure Vessel Wall of the PM-2A Reactor (2.4-in. A350-LF3 Ring Forging)

Longitudinal Specimens Location	Test Temp. (°F)	Energy (ft-lb)	Shear (%)
650°F, 168 hr			
1/4 t	50	34	30
1/4 t	120	51	70
1/4 t	100	45	40
3/4 t	50	50	40
3/4 t	80	59	65
3/4 t	120	83	85
3/4 t	20	44	40
3/4 t	0	22	25
Reannealed additional 168 hr, 750°F			
1/4 t	-80	27	15
1/4 t	-40	43	40
1/4 t	120	131	100
1/4 t	-60	34	25
1/4 t	-100	16	10
1/4 t	0	86	70

Tension Test Data

Longitudinally oriented tensile specimens (Fig. 2), with gage diameters of 0.2520 in. machined from both the outer and inner radius of the PM-2A vessel steel, were evaluated at the -20°F failure temperature and at the +72°F temperature imposed during acid sharpening of the artificial defect. The results are summarized in Table 4.

The general test behavior for the irradiated, PM-2A tensile specimens consisted of an upper yield point rapidly falling to the lower yield-point (0.2% offset) level, which was quite well defined in each case. A comparison of the reduction in area and elongation values of the unirradiated versus the irradiated condition reveals that neither was

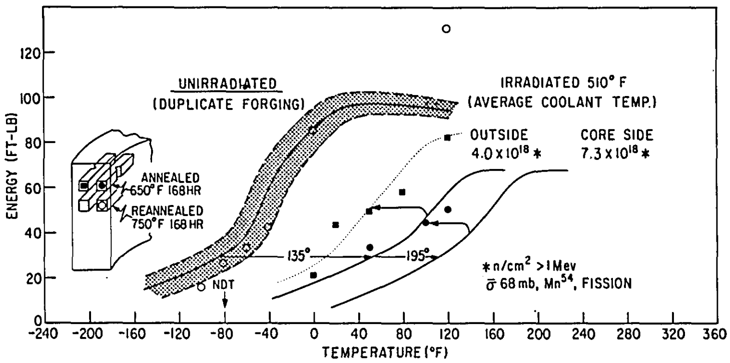

Fig. 5 - Notch ductility characteristics of PM-2A ring-forging steel after postirradiation heat treatment. Closed circles and squares represent specimens heat treated for 168 hr at 650°F; recovery of unirradiated-condition properties relative to the sister forging are between 25 and 35%. Open circles indicate specimens reheat treated for an additional 168 hrs at 750°F; recovery of essentially 100% of the sister-forging unirradiated-condition properties is indicated.

Table 4
Irradiated-Condition Tensile Properties of the Pressure Vessel Wall of the PM-2A Reactor (2.4-in. A350-LF3 Ring Forging)

Specimen Location in Vessel	Fluence (n/cm² >1 Mev)	Temp. (°F)	0.2% Yield Point (psi)	Tensile Strength (psi)	Uniform Elongation (%)	Reduction of Area (%)	Elongation per 1 in. (%)	Hardness (R_B)
Not irradiated*	–	+75	58,600	82,200	†	68.6	33.0	†
1/4 t	7.3×10^{18}	-20	97,620	111,700	7.0	61.7	†	96
3/4 t	4.0×10^{18}	-20	90,450	106,100	10.1	67.3	29.8	94.5
1/4 t	7.3×10^{18}	+72	92,200	105,400	6.2	60.7	24.8	96
3/4 t	4.0×10^{18}	+72	84,200	99,060	7.8	65.5	28.2	94

*Average of six specimens from a sister ring forging.
†Not determined

significantly affected by the irradiation. On the other hand, both the yield and tensile strengths increased markedly with neutron exposure, the increases being generally in line with irradiation data available for the sister forging (6). An overall assessment of the tensile properties, then, would suggest that the vessel steel was not significantly damaged with respect to its ability to accept and accommodate the test loading stresses; in fact, the increased yield strength may have enhanced its ability to accommodate these stresses.

NEUTRON DOSIMETRY AND SPECTRUM

The mechanical properties data shown in Figs. 3 to 5 are referenced to neutron fluences in terms of n/cm^2 greater than 1 Mev, assuming a fission-spectrum distribution of neutrons and a fission-spectrum-average activation cross section of 68 mb for manganese-54 (Mn^{54}). Since neutron-activation detectors or wires were not available for analysis of the vessel lifetime flux and fluence, Mn^{54} induced into the iron matrix of the A350-LF3 steel forging was employed for the fast neutron flux determinations, and cobalt-60 (Co^{60}) induced into the alloyed cobalt was employed for 2200 m/sec thermal neutron flux. For the 1/4-t specimen location, the thermal neutron flux was about twice that of the flux > 1 Mev; for the 3/4-t location, the thermal flux was about 80% of the flux greater than 1 Mev. The sample steel used for dosimetry purposes, 95.98% iron and 0.01% cobalt, was irradiated for a total of 5.16×10^7 sec at an average power level of 5.8 Mw ending July 9, 1963 (7).

The highest neutron flux and fluence value measured from drillings came from the region of the artificial defect adjacent to, but not in, the stainless-steel cladding. Based on the above-noted fission-spectrum parameters and assumptions, the flux was 1.85×10^{11} n/cm^2-sec, and the fluence at this point was 9.5×10^{18} n/cm^2. Drillings were taken at ten locations across and around the periphery of the slab of steel from which the four sections were taken (Fig. 1). The material analyzed was between 1/4 to 1/2 in. from the exterior wall of the vessel. The fluences varied by ±13% from the average indicating that the total 11-in. height and 24-in. length of the stock material for the metallurgical specimens received an exposure to neutrons more uniform than that achieved in many experimental irradiations.

The distribution of neutrons as a function of energy levels has been calculated for several locations in the PM-2A reactor by Ulseth, Yoshikawa, and coworkers at Battelle-Northwest Laboratory using a transport theory Sn code, Program S (8). A listing of arbitrary unit fluxes by energy and lethargy groups for the interface of the pressure vessel wall and stainless-steel cladding (PVW/SS Clad), the 1/4-t location (1.5 in. from PVW/SS Clad) and the 3/4-t location (1.75 in. from PVW/SS Clad) are given in Table 5 and are plotted as histograms in Fig. 6. The 1/4-t spectrum has been omitted for best visibility, since it lies so close to the other spectra. Also, the trend of the peak flux, occurring at progressively lower energies from the PVW/SS Clad to the 3/4-t location, is maintained by the 1/4-t spectrum.

Calculation of the flux spectrum of the PM-2A permits determination of spectral-averaged cross sections ($\bar{\sigma}$) for iron activation, which will yield fluences of neutrons of energies greater than a selected lower energy limit (E_L), such as 1 or 0.5 Mev, according to (9) the relationship

$$\bar{\sigma} = \frac{\int_0^\infty \sigma_{Fe}\,\phi(E)\,dE}{\int_{E_L}^\infty \phi(E)\,dE}, \tag{1}$$

where, σ_{Fe} is the differential cross section for iron as constructed by Helm (10) (Table 5) and $\phi(E)$ is the flux spectrum. The spectral-averaged cross sections for iron activation at the three locations calculated are given in Table 5. These cross sections were used to adjust the fission-spectrum fluence values shown in Figs. 3 to 5, using techniques which have been described (11). It is pointed out that Helm's 82.6-mb fission-spectrum cross section (10) rather than the 68-mb fission-spectrum cross section has been used as a base for determining the calculated spectrum cross sections. Accordingly, Table 6 is a listing of 68-mb-based fission-spectrum and adjusted (82.6-mb-based) calculated

Table 5
Calculated Neutron Flux (Arbitrary Units) per Energy Group for Three Locations in the PM-2A Reactor*

Energy Group Structure		Calculated Relative Neutron Flux at Three Locations (n/cm^2-sec)			Average Cross Section, $\overline{\sigma}$, Fe^{54}(n,p) Mn^{54} (barns)
Lethargy (u)	Lower Energy Limit, E_L (ev)	PVW/SS Clad 47.6-48.31 cm†	1/4 t 49.01-49.72 cm†	3/4 t 51.84-52.55 cm†	
0.25	7.79×10^6	6.951×10^{-7}	5.840×10^{-7}	3.371×10^{-7}	0.450
0.50	6.07×10^6	2.236×10^{-6}	1.871×10^{-6}	1.071×10^{-6}	0.495
0.75	4.72×10^6	3.685×10^{-6}	3.077×10^{-6}	1.750×10^{-6}	0.450
1.00	3.68×10^6	4.025×10^{-6}	3.351×10^{-6}	1.924×10^{-6}	0.295
1.25	2.87×10^6	4.383×10^{-6}	3.666×10^{-6}	2.142×10^{-6}	0.195
1.50	2.23×10^6	6.450×10^{-6}	5.320×10^{-6}	3.067×10^{-6}	0.110
1.75	1.74×10^6	5.405×10^{-6}	4.665×10^{-6}	2.842×10^{-6}	0.015
2.00	1.35×10^6	5.278×10^{-6}	4.700×10^{-6}	3.019×10^{-6}	—
2.25	1.05×10^6	4.649×10^{-6}	4.322×10^{-6}	2.910×10^{-6}	—
2.50	8.21×10^5	4.491×10^{-6}	4.332×10^{-6}	3.078×10^{-6}	—
2.75	6.39×10^5	4.739×10^{-6}	4.465×10^{-6}	3.039×10^{-6}	—
3.00	4.89×10^5	4.279×10^{-6}	4.186×10^{-6}	3.032×10^{-6}	—
3.25	3.88×10^5	2.442×10^{-6}	2.162×10^{-6}	1.419×10^{-6}	—
3.50	3.02×10^5	2.658×10^{-6}	2.259×10^{-6}	1.420×10^{-6}	—
3.75	2.35×10^5	2.333×10^{-6}	1.990×10^{-6}	1.249×10^{-6}	—
4.00	1.83×10^5	1.984×10^{-6}	1.549×10^{-6}	8.467×10^{-7}	—
6.75	1.17×10^4	1.355×10^{-5}	1.130×10^{-5}	6.708×10^{-6}	—
9.00	1.23×10^3	8.162×10^{-6}	6.587×10^{-6}	3.724×10^{-6}	—
11.50	1.01×10^2	8.974×10^{-6}	6.629×10^{-6}	3.122×10^{-6}	—
16.50	0.683×10^2	2.032×10^{-5}	1.510×10^{-5}	6.774×10^{-6}	—
∞	0.0	3.040×10^{-4}	2.148×10^{-4}	8.685×10^{-5}	—
	$\Phi > 1$ Mev	3.770×10^{-5}	3.242×10^{-5}	1.968×10^{-5}	
	$\Phi > 0.5$ Mev	5.032×10^{-5}	4.454×10^{-5}	2.821×10^{-5}	
	$\overline{\sigma}_{cs} > 1$ Mev	159 mb	154 mb	146 mb	
	$\overline{\sigma}_{cs} > 0.5$ Mev	119 mb	112 mb	102 mb	

*Spectral calculation used 25.59 cm as effective core radius.
†Distance from core center to zone represented by dimensional limits.
‡From Ref. 11.

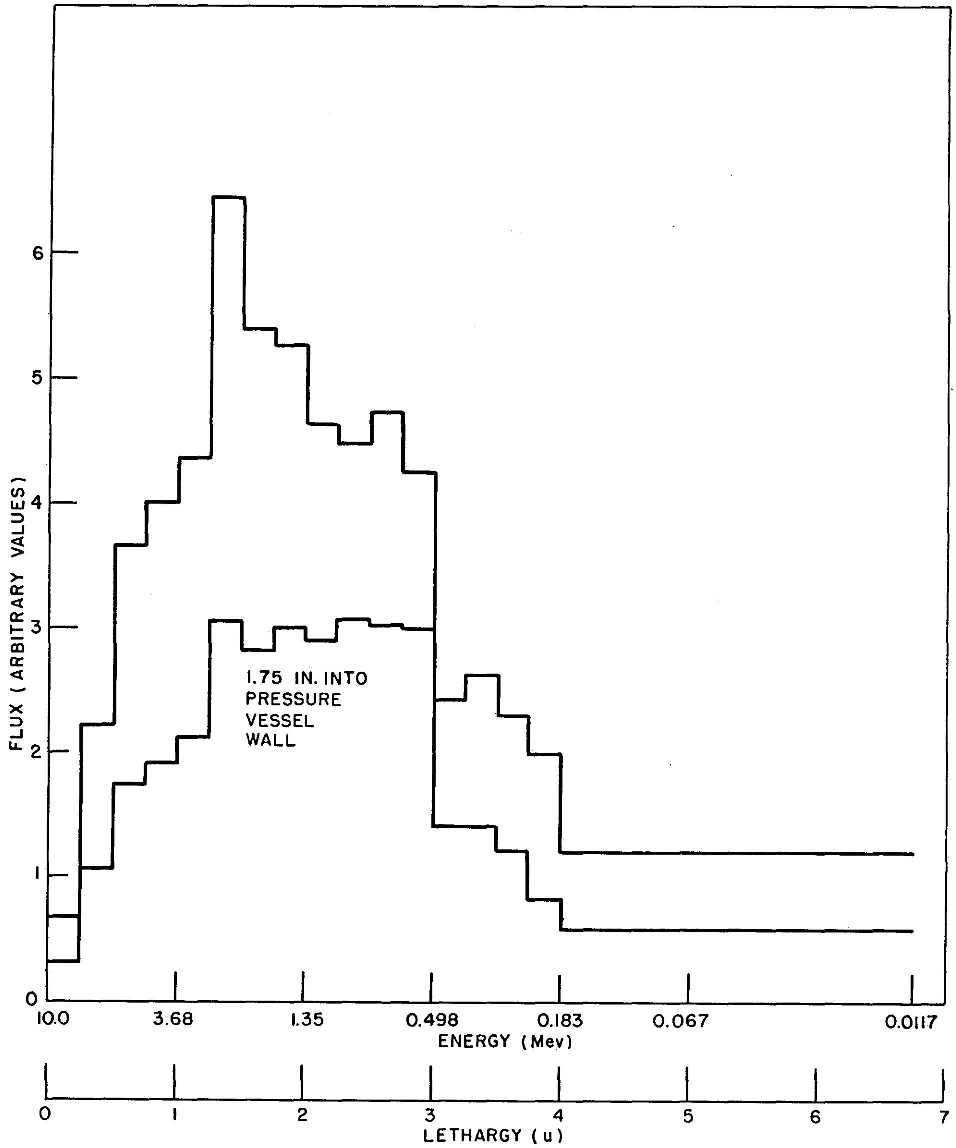

Fig. 6 - Relative intensities of the neutron spectra in the PM-2A reactor at the interface of the pressure vessel wall/stainless-steel cladding and at the 3/4-position located 1.75 in. into the vessel wall (from inside to outside)

spectrum fluence values for neutrons greater than 1 Mev and greater than 0.5 Mev for the notch ductility values shown in Fig. 3. Included in the table are fluences corresponding to notch ductility values determined from irradiations of the PM-2A sister forging in test reactors (12).

The data of Table 6 are plotted in Fig. 7 as referenced to the NRL trend band for irradiations less than 450°F. It is apparent, and probably quite fortuitous, that the open triangles representing the fission-spectrum assumption criterion for determining fluences, describe a satisfactory trend pattern for the range of fluences investigated. This would suggest that estimates of transition temperature increases can be made in terms

Table 6
Notch-Ductility and Neutron-Fluence Data for Irradiation of PM-2A-Reactor Steel

2.4-in. Ring Forging Source	Reactor Location	Temp. (°F)	30 ft-lb Trans. Temp. Increase (°F)	Φ_f* > 1 Mev	$\bar{\sigma}_{cs}$† > 1 Mev (mb)	Φ_{cs}† > 1 Mev	$\bar{\sigma}_{cs}$† > 0.5 Mev (mb)	Φ_{cs}† > 0.5 Mev
Sister	LITR C-55	490	270	1.4×10^{19}	84.9	1.62×10^{19}	57.3	2.40×10^{19}
Sister	LITR C-18	510	320	2.7×10^{19}	96.4	2.75×10^{19}	65.2	4.06×10^{19}
Sister	LITR C-55	500	365	2.9×10^{19}	84.9	3.35×10^{19}	57.3	4.96×10^{19}
PM-2A	PM-2A 1/4 t	510	195	0.73×10^{19}	154	0.47×10^{19}	112	0.64×10^{19}
PM-2A	PM-2A 3/4 t	510	135	0.40×10^{19}	146	0.27×10^{19}	102	0.39×10^{19}

*Based on fission-spectrum-averaged $\bar{\sigma}$ = 68 mb.
†Based on fission-spectrum-averaged $\bar{\sigma}$ = 82.6 mb.

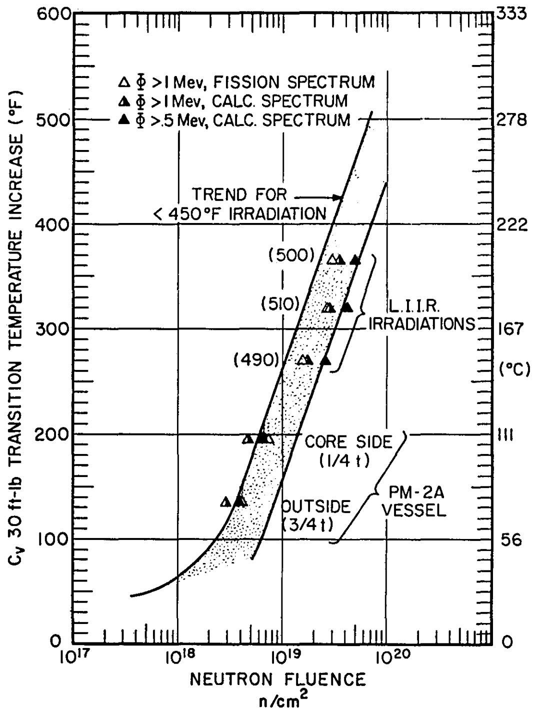

Fig. 7 - Transition-temperature increases (indexed to Charpy-V 30-ft-lb level for a sister ring forging) of the PM-2A-reactor pressure vessel A350 LF-3 forging steel versus neutron fluence determined by assuming a fission spectrum (open triangles) and calculated spectra (partially and completely closed triangles). The Low Intensity Test Reactor (LITR) data were from irradiations at the indicated temperatures using the sister forging as specimen material.

of a fission spectrum with a reasonable degree of assurance, if an acceptable dosimetry measurement can be made.

Note, however, that the calculated spectrum fluences for the PM-2A reactor are lower in value than the corresponding fission-spectrum fluences; conversely, the calculated spectrum fluences from the test reactor irradiations are higher than the fission-spectrum fluences. Note that the comparison shown in Fig. 7 is from the irradiation of two very similar forgings of the same steel heat at essentially the same temperature. This suggests that the differences in the neutron flux spectra of the PM-2A and the test reactor lie at the base of the disagreement.

It is not the intent of this report to fully discuss the relative damaging effect of the neutrons within each of the two types of neutron spectra concerned. It is worth documenting these differences, however, and presenting a thesis to possibly resolve them. Figure 8 shows the PM-2A 1/4-t spectrum plotted versus the Watt fission spectrum; Fig. 9 shows the Low Intensity Test Reactor core lattice C-18 spectrum plotted versus the Watt fission spectrum. The spectra have been plotted by equating the product of the flux and the fission-averaged cross section for flux greater than 1 Mev in the fission spectrum, with the product of the flux and spectral-averaged cross section for flux greater than 1 Mev in the calculated spectra.

Several features of these plots are quite interesting. As anticipated, the PM-2A spectrum is less populous than the C-18 spectrum on an overall basis. However, the fluxes of energies greater than 4.72 Mev (0 to 0.75 lethargy) in the PM-2A spectrum are significantly higher than those in the same energy groups of the C-18 spectra. Also, the large population of neutrons of energies lower than 1.74 Mev (1.75 to 5+ lethargy) in the C-18 spectrum relative to the PM-2A spectrum do not affect the iron activation in any way (see response curves in the upper portion of Figs. 8 and 9). From a brief review of other reactor spectra, both power and test, it appears that the examples shown in Figs. 8 and 9 may well be general cases rather than unique exceptions.

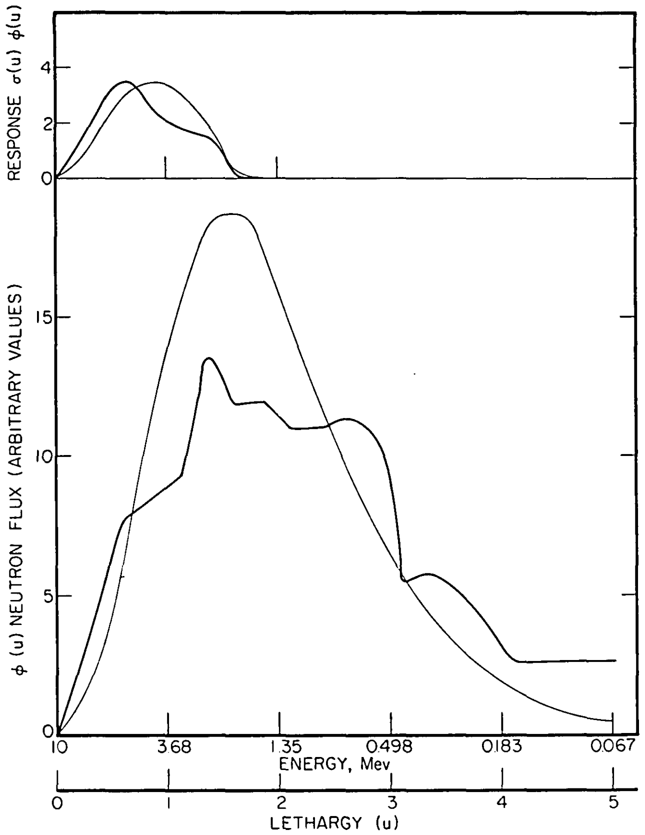

Fig. 8 - Neutron spectrum of the PM-2A reactor 1/4-thickness position (heavy curve, located 1/2-in. into the inside surface of the vessel wall) in relation to the Watt fission spectrum (light curve). The spectra are plotted by equating the product of flux and average cross section per unit energy in the fission spectrum to that for the calculated spectrum. The upper curve shows the response of the flux per unit energy times the activation for the Fe^{54}(n,p)Mn^{54} reaction as interpreted by Helm (10).

Turning again to Fig. 7, the plot of embrittlement data versus neutron fluence values recall that the PM-2A calculated spectrum fluences are smaller than those from the fission spectrum, while the calculated spectrum C-18 fluences are larger in value. It would then appear, on the surface, that more embrittlement is seen for a given fluence of neutrons of energies greater than 1 Mev in a power reactor than in a test reactor. Since this is contradictory to nature, it would appear that the concept of equal weight given to neutrons of all energies greater than 1 Mev will have to be replaced by one which gives more weight to neutrons of energies greater than about 4 to 5 Mev and less weight to those between approximately 0.5 and 1 Mev.

DISCUSSION

Figure 4 shows that the remaining ligament of A350-LF3 steel adjacent to the artificial defect in the PM-2A had Charpy-V 30 ft-lb transition temperatures of about +115°F

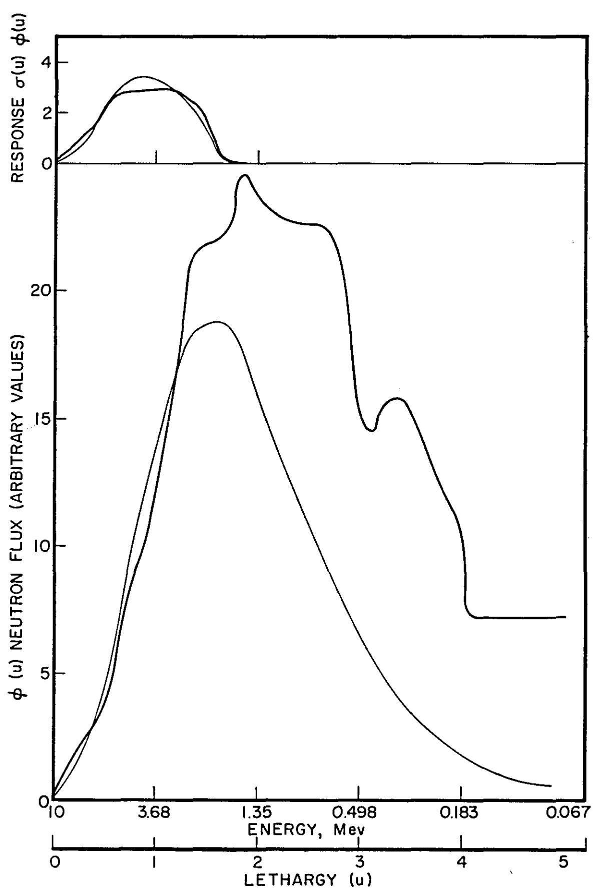

Fig. 9 - Neutron spectrum of the Low Intensity Test Reactor (LITR) core lattice C-18 position (heavy curve) in relation to the Watt fission spectrum (light curve). The spectra are plotted by equating the product of flux and average cross section per unit energy in the fission spectrum to that for the calculated spectrum. The upper curve shows the response of the flux per unit energy times the activation of the $Fe^{54}(n,p)Mn^{54}$ reaction as interpreted by Helm (10).

and higher. Thus, the +72°F acid-sharpening treatments and the -20°F failure (3,4) clearly were accomplished at temperatures below NDT, the region for determination of valid linear elastic fracture mechanics (K_{Ic}) values for crack stress intensity analyses. At this time, however, K_{Ic} values for the PM-2A vessel steel have not been determined. Available mechanical properties data do, however, permit an analysis of the PM-2A failure in reference to the fracture analysis diagram (FAD) procedures of Pellini and Puzak (13,14).

The temperature, stress, and flaw size are three factors on which the FAD is based. The previous paragraph indicates that both the acid-sharpening treatment and failure events occurred below the NDT temperature for most of the vessel wall thickness in the high, irradiated region. (It is to be remembered at this point that actual NDT values have not yet been developed and that Charpy-V 30-ft-lb values are being substituted.)

The vessel hoop stress can be readily determined to a close approximation, assuming the stress in the vicinity of the flaw to be given by the hoop stress, by considering the relationship, $\sigma_h = pr/t$, where p is the applied pressure in psig, r is the vessel radius to the PVW/SS Clad interface, and t is the vessel wall thickness. For the specific pressures of interest, the values in Table 7 result.

The flaw, being the artificial defect in this case, was found to be quite sharp in the acid-treated lower half of the machined defect. The degree of sharpness approached that of a natural or fatigue crack, and there were also numerous secondary cracks visible in the same region which were also produced by the corrosive, embrittling attack of the acid (15).

Table 7
Conversion of PM-2A Test Loop Pressure
Into Vessel Hoop Stresses

Gage Pressure (psig)	σ_h (psi)	Percentage of 0.2% Yield Strength at Temperature (°F)
4000	31,600	34 at 72°F
4475*	35,400*	36 at -20°F*
5000	39,500	43 at 72°F

*Failure conditions; other conditions represent tests.

The flaw size at vessel failure can be seen to compare favorably with the 1/4 to 1/2 (Table 7) yield stress range which is predicted by the FAD. Thus, the combination of a large, sharpened flaw, a pressurizing situation, and a temperature below NDT were, as would be predicted, sufficient to trigger the running brittle crack which fractured the vessel. Referring to the FAD, the failure occurred at a stress level wherein flaws about 1-ft long could be expected to effect fracture. Clearly, a flaw of only several inches would not be expected to fracture the vessel based on the material properties developed and presented in this report. In essence, the conditions at failure (though not perfectly known) were in general accordance with those projected by the FAD.

CONCLUSIONS

1. The steel in the pressure vessel wall of the PM-2A reactor was significantly embrittled by neutron irradiation, but the tensile properties showed only slight deleterious changes in ductility coupled with marked strength increases.

2. The increase in the ductile-brittle transition temperature of steel located throughout the vessel wall compares well with increases developed from test-reactor irradiations when all data are plotted using an assumed fission-spectrum neutron distribution.

3. Neutron dosimetry data from locations across the thickness of the vessel wall present a linear relationship when plotted on a semilog scale. It should, therefore, be possible to accurately estimate through-thickness embrittlement and neutron dosimetry in pressure vessels of reactors provided a valid dosimetry value can be obtained from the vessel itself or from surveillance data.

4. Greater stresses were required to fracture the vessel than would be encountered during normal service. Even with this higher stress, a very large flaw also was required to effect the PM-2A fracture.

5. The conditions for failure of the PM-2A reactor vessel are in agreement with the generalized FAD, using the irradiated-condition yield stress as the reference rather than the unirradiated-condition yield stress.

6. Documentation of the actual irradiated-condition mechanical properties of a reactor pressure vessel, which compare favorably with trends developed in experimental test reactors, affords proof that these values rather than the unirradiated-condition properties should be used in assessing the safety margins available for each individual reactor situation.

7. The fracture experience with the PM-2A reactor suggests increased confidence in the overall safety of reactors.

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13. ABSTRACT

Following the pressurization-to-failure testing of the PM-2A reactor pressure vessel, several sections of steel were removed from the vessel wall in a region adjacent to the artificial defect. Charpy V-notch and tension test specimens machined from one of these sections have been evaluated. The irradiated-condition 30 ft-lb transition temperatures for the 1/4-thickness (nearest to the core) and 3/4-thickness locations in the vessel wall were +115°F and +55°F, respectively, for measured fission-spectrum fluences of 7.3 and 4.0×10^{18} n/cm^2 (>1 Mev). The 1/4-thickness properties and fluence most nearly represented those at the tip of the artificial defect. The 0.2% yield strength for the 1/4-thickness location was 97,620 psi at -20°F (failure temperature) and 92,200 psi at +72°F (temperature at time of acid-sharpening treatment of artificial defect). Significant uniform elongation, reduction of area, and elongation per 1 in. were retained by the steel. An assessment of the stress, temperature, and flaw-size conditions for the PM-2A failure, as indexed by the irradiated-condition mechanical properties, indicates that the failure is in agreement with the generalized fracture analysis diagram.

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